Approved For Release
2009/08/17 :
CIA-RDP88-00904R000100100

Third United Nations
International Conference
on the Peaceful Uses
of Atomic Energy

coded 11/20/64

A/CONF.28/P/320
USSR
May 1964
Original: RUSSIAN

Confidential until official release during Conference

PHYSICAL AND OPERATIONAL CHARACTERISTICS OF REACTOR SM-2

S.M. Feinberg, N.A. Dollezhal, E.D. Vorobjov, V.A.Tsikanov,
I.Y. Emeljanov, V.M. Grjazev, A.S. Kochenov, Y.M. Bulkim,
V.I. Ageenkov, P.G. Averjanov

Introduction

The reactor SM-2 is designed for carrying out various scientific investigations in the field of nuclear physics, solid-state physics, physical metallurgy, radiation chemistry, reactor physics and engineering and many others.

The reactor core is a parallelepiped measuring 42x42x25 cm^3 with an inner cavity of 14x14x25 cm^3 which serves as a neutron trap. The corners of the parallelepiped measuring 7x7x25 cm^3 are occupied by the shim rods. The core is formed by 7x7x25 cm^3 fuel assemblies and is surrounded by a reflector made from berillia blocks interspaced by water. The fuel assembly contains 54 plate-type fuel elements, 0.8 mm thick, separated by 1.65 mm spaces filled by water.

To carry out experimental work the reactor is provided with 5 horizontal, 1 slanting and 18 vertical channels. The physical characteristics of the reactor and its main parameters were described in Ref.1, its main constructional features, in Ref.2. The reactor SM-2 is the first research reactor in the world using intermediate neutrons and water moderator.

The construction of the reactor was completed in 1961. The physical start-up was effected in October, 1961. Since November 1o62 the reactor has been operating at rated parameters.

The maximum power achieved in the reactor is 55 MW.

During the time the reactor was operated at decreased power and, especially, during operation of the reactor at rated parameters a large programme of research was carried out including the study of the reactor itself and of its separate parts and irradiation of various substances in the experimental channels.

Besides, operation of the reactor during this time has made it possible to ascertain the performance characteristics of its main elements.

The idea to construct the research reactor SM-2 employing intermediate neutrons as well as the design of the reactor were suggested by S.M. Feinberg who was also in charge of the entire work involved in the construction of the reactor. The reactor construction was worked out under the guidance of N.A. Dollezhal and Y.M. Bulkin. The experimental research at the initial stage of the project was performed by E.D. Borobjov, V.B. Klimentov and V.M. Grjazev. Some physical calculations were made at this stage by I.K. Levinal and N.Y. Lashchenko who prepared the mathematical programme of reactor calculation in P_1 approximation. At the second stage of the project the experimental investigation of the reactor physics was continued with collaboration of V.A. Tsikanov and others. V.A. Tsikanov who was the closest assistant of the scientific advisor of the project contributed an appreciable share to the development of the technological scheme of the reactor and supervised the construction and operation of the reactor at the last stage. The control and safety system of the reactor was developed under the guidance of I.Y. Emeljanov. Beginning from the second stage the physical calculations were mainly done by A.S. Kochenov. V.I. Ageenkov suggested the construction and the manufacturing techniques for the fuel elements of the reactor core. P.G.Averjanov and other members of the operational staff took an active part in mastering the operation of the reactor SM-2.

The development, construction and operation of the reactor SM-2 were carried out with the participation of a large team of researchers, designers, engineers and workers to whom the authors express their deepest appreciation.

Chapter 1. Physical Characteristics of the Reactor

The linear dimension of reactor SM-2 are commensurate with the neutron migration length, the reactor core and the reflector have complex geometry. Therefore when designing the reactor it was rather risky to relay completely on the physical calculation and the main emphasis was placed on the experiments.

The initial experiments were described in Ref.1. These mainly included the experiments with homogeneous cores containing constructional materials in addition to the uranium and moderator. The number of critical experiments with circular cores and water reflector was insufficient. Due to uranium shortage it was not possible, during these experiments, to introduce the required amount of constructional materials into the core. Several critical experiments were made with a homogeneous core and beryllia reflector which covered only a small part of the side surface of the core.

On the basis of this apperently incomplete information it was expected that the critical charge in the reactor SM-2 would be

320

within 6.8 - 7.3 kg. The loss of reactivity due to the experimental channels was disregarded. However, the subsequent experiments which completely simulated the core and the reflector of the reactor SM-2 showed that the critical charge was much larger.

The experiments were accompanied by the development of a suitable computation technique. It was established that the calculation curried in P_1 approximation (12 energy groups) [1] of the neutron flux distribution over the core and the central water cavity is in good agreement with the experimental data. However, the value critical charge is in good agreement with the experimental data only in the case of one-dimensional geometry of the core and reflector. It became obvious that the physical calculation of some core configurations could be made only in two-dimensional approximation.

The calculation of the neutron spectrum in the core indicated that the bulk of neutrons (approx. 90%) are absorbed at energies below 10 ev. The age of thermal neutrons is 35 cm^2 while the energies within 10 - 0.1 ev account for as little as about 2 cm^2. Therefore the neutron leakage from the reactor is determined by energies higher than 10 ev.

Owing to this the following two-group model is suitable for neutron-physical calculation: fast neutrons are not absorbed but undergo considerable migration; only slow neutrons (with energies below 10 ev) whose migration is small are absorbed.

The mathematical programme used for solution of reactor equations in two-group two-dimentional approximation is described in Ref.3.

The reactor calculations were made both for the plane and cylindrical geometries. In the plane geometry the neutron leakage in the vertical direction was accounted for by assuming that the group cross-sections of absorption Σ_i are equal to

$$\Sigma_i = \Sigma_i^{*} + \varkappa_z^2 D_i \qquad (1)$$

where Σ_i^{*} - absorption cross-section of the i-th group in infinite medium;

D_i - diffusion coefficient of the i-th group;

$\varkappa = \left(\frac{\pi}{H+2\delta}\right)$

H - core height;

δ - reflector saving.

The neutron leakage through the vertical gas channels was accounted for in the following manner: all macroscopic sections within the channel were assumed to be zero and the coefficient of diffusion

$$D_i = \frac{d_r}{3} + \frac{\lambda_{tri}}{3} \tag{2}$$

where d_r - the hydraulic diameter of the gas **channels;**

λ_{tri} - transport length of the i-th group in the space surrounding the channel.

The effect of the horizontal channels on the reactivity was accounted for by means of experimental corrections.

In cylindrical geometry only the neutron flux distribution along the reactor height was calculated. The effect of the experimental channels was not considered.

A two-group two-dimensional analysis was used to calculate the reactivity for the cores of various configurations, the reactivity worths of the shim rods, changes of the reactivity due to the burnout of U-235, distribution of the fast and slow neutron fluxes. The calculations show satisfactory agreement with the experiments.

1. Critical Charge

The critical charge was investigated for various core configurations: with a central water cavity, with a beryllium inserts installed in the water cavity, with a central cavity filled with beryllia, etc.

The experiments were carried out on a special physical installation and directly inside the reactor vessel. The criticality was reached by gradually filling the preassembled core with water. The results of some experiments which aimed at finding the critical charge of the reactor are given in Table I.

The critical mass of the real system with a central water cavity without experimental channel was found to be equal to 11 kg against the expected value of 6.8 - 7.3 kg mentioned above.

The experimental channels increase the critical mass to 13.5 kg. To decrease the value of critical mass beryllium inserts were installed into the neutron trap which lowered the critical mass to 8.6 kg and to 11.2 kg without and with experimental channels, respectively.

The two-group two-dimensional physical analysis gives somewhat decreased values of critical charge. However, the discrepancy with the experimental data does not exceed 10%.

Table II shows the values of reactivity which appears as a result of replacement of a beryllia assembly by the fuel assembly for the core configuration shown in Fig.9,g.

As would be expected these results indicate that the effect of the fuel assembly on the reactivity depends on the core configuration,

320

on the location of the assembly and on the presence of experimental channels near the assembly.

2. Reactivity Control

The worths of the control rods were found by the reactor riding-up period. The reactivity was calculated by the inhour formula. On the basis of special calculations the ratio $\frac{\beta_{eff}}{\beta}$ was adopted to be equal to 1.4 for the core consisting of 20 assemblies or to 1.3 for the core consisting of 28 assemblies.

The worth of the shim rods located at the corners of the core for three core configurations is shown in Table III.

The tabular data indicate that the reactivity of four shim rods is equal to about 4.5%. No interference between the rods has been detected. However, this was not sufficient for normal operation of the reactor. Therefore to augment the control and safety system four safety rods were placed in the beryllium inserts installed inside the neutron trap in addition to four shim rods. The safety rods consist of a beryllium expeller and cadmium tube (1 cm dia.) filled with water. During operation of the reactor the absorbing part of the safety rods is located above the core and, consequently, does not perturb the neutron flux. The worth of one rod is 1%. Considering the interference the worth of four rods is equal to 2.8%.

It was found that four shim rods installed inside the reflector mass had low reactivity. Therefore they were removed from the reactor and the channels were released for experimental purposes.

The total reactivity of the shim and safety rods with 28 fuel assemblies loaded into the core is 6.6%.

3. Effect of Experimental Channels on Reactivity

The effect of experimental channels on the reactivity was studied on a physical experimental installation. It has been established that the materials with a small absorption cross-section (aluminium, lead, beryllia, beryllium), loaded into the water-filled central channel increased the reactivity. Beryllium and beryllia are especially effective in this respect. Thus, for example, the beryllium inserts installed in the neutron trap between the central channel and the core, raise the reactivity by ~5%.

Six 1 cm. dia. cylindrical slugs of 2% enrichment with a total

320

U-235 loading of 18 gr. uniformly arranged over the cross-section of the central channel raise the reactivity by $\sim 1\%$. Ten plate type fuel elements of the reactor SM-2 spaced by 2.45 mm bring the reactivity up by $\sim 2\%$.

The effect produced on the reactivity by various materials placed in the peripheral vertical and horizontal channels is illustrated in Tables IV and V. The tabular data are given for core configuration shown in Fig.9,g.

Distant peripheral channels produce no appreciable effect on the reactivity.

The total reactivity loss due to the presence of experimental channels in the external reflector amounts to $\sim 4\%$.

4. Spatial-Energy Distribution of Neutrons

The experimental study of the neutron spectrum and the spatial distribution of the neutron flux within the reactor volume was made with the help of small-size pulse chambers containing U-235 and also with uranium, indium and gold indicators. The diameter of the fission chambers (1.4 mm) made it possible to make the measurements in the gaps between the fuel plates. The accurate shifting and registering of the fission chamber were accomplished by means of a special measurement bridge. The fission density of U-235 in the experimental channels was measured by means of chambers with a large diameter (up to 5 mm).

The results of measurements of $Q=\int \varphi(u)\sigma^{f}(u)\,du$ are shown in Fig.1.

The solution of the kinetic equation points out to considerable blocking of the neutron flux in the fuel elements. The ratio of the maximum value in the water space Q_{H_2O} to the average value in the fuel element core $\bar{Q}_{bl}$ is equal to about 1.3. The ratio $Q_t/\bar{Q}_{bl}$ (where Q_t corresponds to the maximum flux in the trap) makes it possible to relate the maximum flux of thermal neutrons to the specific power of the reactor; therefore ratio $Q_t/\bar{Q}_{bl}$ rather than Q_t/Q_{H_2O} is of prime interest.

Fig.1 gives the calculated curves of the fission density distribution measured by a chamber with U-235. The dots show the results of the experiment in the neutron trap multiplied by 1.3.

The experimental data indicate that the non-uniformity of heat release in the core is ~ 3.

The radial density distribution of epicadmium neutron activation as well as cadmium ratios for indium and gold are shown in Fig.2. The experiments were made with 0.1 gr/cm^2 thick indium indicators and 0.2 gr/cm^2 thick gold indicators.

The neutron fluxes were calculated in various approximations. The spacial-energy distribution of slowing-down neutrons and the fission density curves measured with U-235 chamber were calculated in P_1 approximation.

The fission density curves were also calculated in a two-group two-dimensional approximation.

The spectrum of slow neutrons for a homogeneous medium was calculated with the help of equation (4).

$$\Phi(Z)=\frac{Ze^{-Z}}{1+\frac{\gamma\Sigma_a}{\xi\Sigma_s}}\left[\int_0^Z dZ'\frac{\xi\Sigma_s Z e^{-Z}+\frac{d}{dZ'}(\xi\Sigma_s Z'e^{-Z'})}{(\xi\Sigma_s Z'e^{-Z'})^2}\int_0^{Z'}\Sigma_a(Z'')\varphi(Z'')dZ''+\mathrm{const}\right] \quad (3)$$

where $Z=\frac{E}{KT}$, T - temperature of the medium

$\Sigma_s(Z)$ - macroscopic scattering cross-section

$\Sigma_a(Z)$ - macroscopic absorption cross-section

ξ - mean logarythmic energy loss at neutron scattering on a free stationary nucleus, γ - half ratio of mean logarythmic square of energy loss to mean logarythmic energy loss at neutron scattering on a free stationary nucleus.

Equation (3) was solved on electronic computer M-20 on the assumption that Σ_s = const. and $\gamma=1$. The law of absorption cross-section variation with energy was adopted in accordance with the absorption cross-section in U-235. The neutron spectrum was calculated for energies from 0 to 2 ev. Simultaneously, the slowing-down flux was calculated along the energi axis $\gamma(E)$, which is equal to the portion of the neutrons absorbed below the given energy, i.e.

$$j(E)=\frac{\int_0^E \sigma_c^s\varphi\, dE'}{\int_0^\infty \sigma_c^s\varphi\, dE'} \quad (4)$$

The temperature of the medium was assumed to be $50^{\circ}C$.

The energy dependence of the neutron flux $\varphi(Z)$ and the slowing-down flux along the energy axis is shown in Fig. 3.

The theoretical U-235 cadmium ratio is equal to 2.44 for capture and 2.47 for fission. The experimental value of the U-235 Cd ratio for fission in the core centre varies within 1.5 - 2.1.

The measurements of the fission density distribution in the experimental channels was conducted in a system shown in Fig.9,g. The vertical channels Nos 2,3,4,5,6,7 and horizontal channels Nos.1,2,3,4,5, were filled with air, vertical channels Nos.1,8,9,10,11,13,14,15 were filled with water. The fission density distribution for U-235 in height was measured in vertical channels Nos.4,12,14 and 15. The results of the measurements are shown in Fig.4.

The data on the fission density of U-235 in the vertical channels at the level of the core centre are summarized in Table VI. During the measurement all four shim rods were inserted into the core up to their centre.

Of much interest is the neutron flux distribution in the central water cavity when various amounts of fissionable materials are brought into the cavity. This was investigated by carrying out a series of measurements the results of which are shown in Fig.5.

Fuel plates of the reactor SM-2 were installed in the channel at various spacings. As shown by the Chart the installation in the channel of 14 plate-type fuel elements spaced by 2.45 mm reduces the thermal neutron flux about 20-fold.

The neutron spectrum in the vertical channels was measured by activating threshold indicators (Au, In, Mg, Ti, Fi, Al). Fig.6 shows the neutron spectrums for several channels.

5. Steady-State Poisoning of the Reactor with Xenon-135

The data on the steady-state poisoning of the reactor with xenon-135 are given in Table VII.

The poisoning was calculated by the formula

$$\rho = \frac{\bar{\sigma}_f^5}{\bar{\sigma}_a^5} \cdot \frac{(\gamma_1+\gamma_2)\sigma_{Xe}\Phi dE}{\lambda_2 + \int \sigma_{Xe}\Phi dE} \qquad (5)$$

where $\gamma_1+\gamma_2$ = 0.064- the total fission yield of iodine and xenon

$\lambda_2 = 2.09 \times 10^{-5}\,sec.^{-1}$- the decay constant of Xe-135

The formula (5) corresponds to the following approximations:

1) The nuclear density of Xe-135 is considered to be constant throughout the core and equal to some value corresponding to the average fission density.

2) Importance of slow neutrons is also constant throughout the core.

3) It is assumed that the spectrum of slow neutrons throughout the core is the same as inside the core.

In the actual fact, however, a neutron flux flash-up is present at the boundary of the core and the neuclear density of xenon in these places may considerably differ from the average value. Besides the importance and spectrum of slow neutrons in these places will differ from the corresponding values in the core mass. These boundary effects were accounted for in accordance with the perturbation theory. The change of the reactivity caused by the reactor poisoning was calculated by the formula

$$\frac{\Delta K}{K} = \theta \rho \qquad (6)$$

Where θ -the portion of the neutrons absorbed in the uranium.

320 Table 7 shows that the reactivity of poisoning measured at a power

of 28.0 MW exceeds that at a power of 33.7 MW.

This contradiction brings out the imperfection of the experimental technique for the reactivity measurement and shows that the measurement error is as large as 0.25%.

At a power of 50 MW the Xe-135 poisoning is close to the limiting value and is equal to $\frac{\Delta K}{K} \simeq 4\%$.

6. Uranium Burn-up and Reloading of Assemblies

During reactor operation the reactivity decreases at a rate of 2.2×10^{-4}%/MWhr. The two-group two-dimensional analysis of the reactivity changes made on the assumption of a uniform uranium burn-up within one assembly gives the result of $(1.7-2.3) \times 10^{-4}$%/MWhr. The same analysis indicates that the replacement of core assemblies uniformly burnt out to 12.5% in the inner row of the core by new assemblies increases the reactivity by 1.3%, similar replacement in the outer row gives a reactivity increase of 1%. With the burnups as large as 25%, these values are equal to 2.8% and 2%, respectively.

7. Temperature Coefficient of Reactivity

The experiments carried out to find the effect of temperature on the reactivity indicate that the reactivity monotonically decreases as the water of the first circuit is heated at constant temperature of the water in the central channel. Conversely, the reactivity increases when the water in the central channel is heated and the temperature of the water in the first circuit remains constant. When the water in the core and in the central cavity is heated simultaneously, the reactivity increases at up to $\sim 30^\circ$C and then drops. The results of these experiments are illustrated in Fig.7.

8. Distribution of Uranium within Assembly

One of the most complicated problems with reactors of the SM-2 type is the correct distribution of the uranium at the core boundary.

The experiments made on the fuel elements of the reactor SM-2 has shown that the fuel elements can operate at heat flux above 10^7 Kcal/m^2hr which exceeds two-fold the maximum heat flux in the reactor. Therefore, the distribution of the uranium in the core aims at increasing the average burnup rather than at decreasing the heat flux.

The calculations show that unless the uranium loading in the assemblies is properly distributed the average burnup of the uranium in the assembly unloaded from the core is $5K_z$ smaller than the permissible value ($K_z = 1.3$ - the non-uniformity coefficient of heat

320

release along the core height.) This means that, for example, if the permissible burnup is 35%. the average burnup will not exceed 6%.

The uranium distribution adopted in the project (the first plate contains 1/3 of uranium, the second 2/3, the third and all subsequent plates, 1) has enabled the burnup to be increased about 1.7 times. This is due to the fact that the first plate having the full uranium charge is the third from the inner boundary of the core where the neutron fluxes are smaller. The plates with a lower uranium content, i.e. with a larger permissible burnup are arranged in the region of higher fluxes.

New distribution (0.25, 0.4, 0.6, 0.8, 1.0) introduced in the course of reactor operation permitted the average burnup to be increased 2.5 times as compared to assemblies with uniformly distributed uranium.

All burnup data given above refer to a core without beryllium inserts. The provision of the beryllium inserts in the neutron trap somewhat decreases the neutron fluxes on the inner boundary and raises the average burnup about 1.3 times.

Therefore the measures that were taken have made it possible to bring the average burnup closer to the permissible one by more than 3 times. However, the difference between the average and permissible burnup is still as large as ~ 2.

The non-uniform distribution of uranium within the block raises the critical charge. However, the charge increases at a slower rate than the average burnup. Thus, for example, the distribution provided for by the project increased the critical charge by about 5% and new distribution, by about 10%. Consequently, proper non-uniform distribution is warranted from the standpoint of uranium consumption. As the uranium is spent the masimum heat flux slowly decreases in value and shifts in the direction of the core mass. Therefore to find the correct distribution the burnup processes at the core edge must be analysed in time. The complete solution of this problem is yet to be found.

Fig.8 shows the redistribution of the thermal load in the fuel element at the edge of the core for the uranium distribution provided for by the initial project.

Chapter II

OPERATIONAL CHARACTERISTICS OF REACTOR

1. Fuel Elements

According to the project the fuel elements of the reactor were to operate at thermal loads of $(5\text{-}6)\times10^6$ Kcal/m^{2}hr. The attainment of such

320

Approved For Release 2009/08/17 : CIA-RDP88-00904R000100100021-7

thermal loads and close arrangement of plate-type fuel elements in the reactor core (the thickness of the fuel element is 0.8 mm, the thickness of the water space, 1.6 mm) have made it possible to obtain a high specific load in the core (q=1600 - 1700 kW/lit.;$q^{max} \approx 4500$ kW/lit.). Though the heat engineering calculations indicated that the fuel elements could operate at such high thermal loads the possibility of attainment of such loads was generally doubted. Besides, it was necessary to check the possible burnup fraction of U-235.

Only the experiment could give the answers to all these questions. Therefore before attempting to operate the reactor at the rated power the fuel elements were tested in channel No.I at a thermal load of 6×10^6 Kcal/m^{2}hr to the maximum U-235 burnup of 28%. The total number of the plate fuel elements tested during the experiment was only six. During operation of the reactor at the rated power the thermal load of 6×10^6 Kcal/m^{2}hr was achieved simultaneously on a large number of the fuel elements while the maximum burnup of U-235 within the reactor core reached, with some fuel elements, as much as 35% and even more.

During reactor operation two fuel assemblies with the U-235 burnup of 2-3% were extracted from the core because of the suspicions as to their hermetical sealing. The small burnup is indicative of the defective sealing. To find out whether it was possible to increase the thermal load, the plate fuel elements were tested for the burnup of U-235 at a thermal load of $(10-11) \times 10^6$ Kcal/m^{2}hr. In these experiments the maximum burnup was as large as 30%. The tests show that the thermal load on the surface of the fuel elements can be increased two times as compared with the value stipulated by the design and only in this way more intensive neutron fluxes can be obtained in the reactor.

Therefore, the reactor operating experience and the experiments proved the serviceability of plate fuel elements for intermediate-neutron reactors operating at high thermal loads.

2. Control and Safety System

After installation of additional rods during the prestart preparations the control and safety system of the reactor was capable of providing the necessary reactivity margin allowing for the loading of additional fuel assemblies to compensate for the burnout of U-235.

During reactor run the safety system provided complete ensurance of safety, the false operations of the system were rare.

There were no major faults in the electronic instruments and electric circuits comprised in the safety system, power measurement and automatic control systems. As a result of preventive and routine maintenance the system was constantly kept in a serviceable condition.

320 After certain improvements were introduced during the start-up

Approved For Release 2009/08/17 : CIA-RDP88-00904R000100100021-7

period the hermetic actuators of the control rods mounted on the reactor cover operated without any malfunctions during the entire service period. In the course of reactor service a minimum of preventive maintenance operations were carried out including the regular measurement of the insulation resistance of electric coils and the checking of adjustment of control units.

Hydraulic actuators were installed to drive four safety rods arranged in the neutron trap. In the course of service the actuators operated trouble-free and required only periodic adjustment of the signalling system.

3. Data on Stability of Various Structural Parts of the Reactor

The problem of material stability is especially important for a reactor with a high neutron flux. In view of this the construction of such a reactor must provide for the maximum replaceability of parts.

In the course of development of the reactor SM-2 the problem of selection of the material for the reflector has many times come under prolonged discussions. However, no experimental data were available on the irradiation by a high integral neutron flux to make the final solution. It was decided to use the beryllia as the first version. In the course of reactor operation the condition of the beryllia is checked by taking the samples from the replaced assemblies and from the shanks of the shim rods. The checking is done within integral fluxes from 10^{20} to $10^{22} n/cm^2$ for fast neutrons ($E \geqslant 1 Mev$). The measurements show that $10^{22} n/cm^2$ is a limiting value of the integral flux for the beryllia.

Therefore at present the replaceable beryllia assemblies are substituted by the assemblies made from metal beryllium.

The effect produced by a fast neutron flux on the beryllia layers in the non-replaceable reflector is not so great and the construction of the reflector incorporates provisions against destruction of the beryllia blocks.

The other highly essential reactor parts operating under severe conditions are the cans of the horizontal channels which pass through the side reflector up to the edge of the core. The bottoms of these cans are irradiated by a fast-neutron flux of $5 \times 10^{14} n/cm^2 sec$ and by γ-rays from the core. Besides they have to stand up to an external pressure of 50 kg/cm^2.

Lengthy operation of the cans results in the considerable radiation damage of the material of their walls. This damage can be kept within permissible limits by periodically replacing the cans. The possibility of such replacement is provided for by the construction.

4 Operation of Accessory Equipment

During reactor service the entire equipment of the cooling circuits and accessory systems operated without any malfunctions. The pumps of the primary and secondary cooling circuits, heat-exchange equipment and cooling tower ensured the diversion of the reactor heat in any season of the year.

The accessory equipment of the experimental circuits likewise operated without any troubles.

Conclusions

The operation of the reactor SM-2 has confirmed the correctness of the main physical and engineering ideas incorporated in the design project:

1. A thermal-neutron flux of about 2.5×10^{15} n/cm^{2}sec. was reached in the trap at a power of 50,000 kW.

2. The fast-neutron flux with an energy above 1 Mev in the core exceeded 10^{15} n/cm^{2}sec.

3. A maximum thermal load of 4.5×10^3 kW per one litre was attained in the core at the non-uniformity coefficient over the core of about 3.

4. The fuel elements remained completely serviceable at a thermal load of 6×10^6 Kcal/m^{2}hr, which is a record-breaking one for the reactor engineering; the operating experience has shown that this load can be doubled. In this case the permissible burnup fraction exceeds 35%.

5. The main engineering assemblies of the reactor-cooling system, reloading system, control and safety system, etc., proved to be completely serviceable.

At present a project is being developed to increase the reactor power. No changes will be made in the reactor vessel, main circuit, auxiliary systems and structures. The reactor power is to be increased from 50,000 kW to 100,000 kW. The core is to use the fuel elements manufactured according to the technology which was developed and tested before. The core height is to be increased to 40-50 cm. The BeO reflector is to be replaced with the reflector made from metal beryllium; the reflector will consist of separate replaceable blocks. To increase the thermal-neutron flux in the horizontal channels water cavities are to be provided in the direct vicinity to the horizontal channels.

The increase of the power of the reactor SM-2 will make it possible to: 320

1. Raise the thermal neutron fluxes in the trap to $(5-8)\times10^{15}$ n/cm^2 sec (depending on the number of the samples irradiated in the core) and fast-neutron fluxes (at energies exceeding 1 Mev) to 2×10^{15} n/cm^{2}sec.

2. Irradiate the samples in the "hard" spectrum of the core.

3. Increase the average burnup of U-235.

REFERENCES

1. S.M.Feinberg, E.D.Vorobyov, V.N.Gryazev, V.B.Klimentov, N.Y.Lyashchenko, V.A.Tsikanov. "An Intermediate Reactor for Obtaining High Intensity Neutron Fluxes." Paper presented to the 2nd U.N.Conference on the Peaceful uses of atomic energy.
2. S.M.Feinberg et al. "Atomnaya Energia", 8, No.6, p.p.493-504(1960).
3. Y.V.Shevelyov, V.K.Saulyev. "Atomnaya Energia", 14, No.2, p.p. 200-205 (1963).
4. N.I. Laletin, "Atomnaya Energia", 14, Vol.5, p.p.458-464 (1963).

320

Table I

Critical Masses for Various Configurations of the Core

Type of system	Condition of channels	Core configuration shown in Fig.9	Number of assemblies[x)]	Weight of U-235, kg	Reactivity, %	
					Experiment	Theoretical
BeO assemblies are installed in central cavity	Channels are open	a	13	8.6	0	
Central cavity is filled with water	Vertical channels are closed by BeO plugs Pb plug is placed in horizontal channel III BeO plug is placed in horizontal channel V	b	20	13.2	0.5	1.8
Central cavity is filled with water	Vertical channels are closed by BeO plugs Pb plug is placed in horizontal channel III BeO plug is placed in horizontal channel V	c	20	13.2	0	1.1
Central cavity is filled with water	Channels are open	d	21	13.9	0.7	0.8[xx)]
Central cavity is filled with water	Channels are open	e	22	14.5	0.7	
Central cavity accommodates water and Be inserts	Channels are open	f	18	11.9	1.0	

Notes: x) The system becomes subcritical when the number of assemblies in reduced by one.

xx) The reactivity has been calculated for a 20-assembly system.

Table II

Reactivities Appearing as a Result of Replacement of BeO Assembly with Fuel Assembly (see Fig.9,g)

Cell	A-2	A-5	Б-1	Б-5	Б-6	B-2	B-5	B-4	E-4
Reactivity,%	0.67	1.05	0.56	1.13	1.08	1.05	1.57	1.29	1.22

Table III

Reactivity of Shim Rods (SR)

Rod	Reactivity, %			
	Experiment			Calculation
	20 assemblies, Fig.9,c	22 assemblies, Fig.9,e	23 assemblies, Fig.9,g	20 assemblies, Fig.9,c
SR I	1.11	0.81	0.81	1.15
SR II	1.13	1.53	1.82	1.15
SR III	1.11	1.35	1.25	1.15
SR IV	1.05	0.77	0.71	1.15

Table IV

Effect Produced on Reactivity by Various Materials Placed in Peripheral Vertical Channels (see Fig.9,g)

Channel conditions	Reactivity variation, % in channels	
	№2 and №3	№4 and №5
Replacement of gas with water	-0.03	-0.08
Channel cladding (steel tube 69 mm dia., wall thickness 3 mm)	+0.04	+0.18
BeO plug 62 mm dia	+0.1	+0.58
Cluster of nickel rods in tight packing		+0.04
Loading of 170 gr. of U-235	+0.23	

Table V

Effect Produced on Reactivity by Various Materials Placed in Horizontal Channels

Channel condition	Reactivity variation, %	
	Channel separated from core by BeO layer 7 cm thick	Channel immediately adjoins to core
BeO plug 62 mm dia., 300 m long	+0.1	+0.43
Water plug 73 mm dia., 260 mm long	+0.04	+0.22
Plug from lead discs 72 mm dia., 8 mm thick with organic glass spacers, total length 262 mm	+0.1	+0.43

Table VI

Distribution of U-235 Fission Density in Vertical Channels at the Level of Core Centre (see Fig.9,g)

Channel Nº	1	4,5,15	2,3,13	12	14	6	9,10	8	7	11
Relative fission density	1	0.11	0.067	0.057	0.045	0.034	0.030	0.027	0.017	0.013

Table VII

Stable-State Poisoning of the Reactor with Xenon-135

Power, MW	Number of assemblies in core	Average power, MW/lit.	Poisoning $\frac{\Delta K}{K}$ %	
			Experiment	Calculation
4.1	20	167	2.03	2.00
28.0	28	818	3.77	3.55
33.7	28	985	3.69	3.70

320

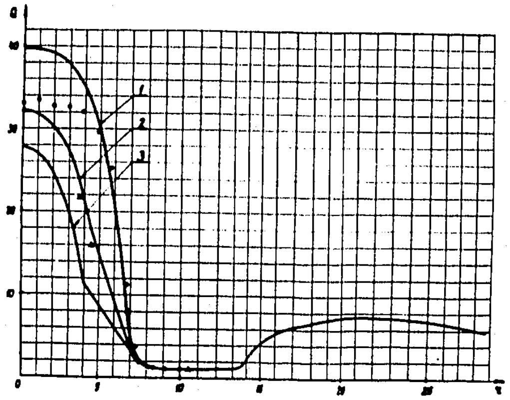

Fig.I. Fission Density Distribution of Uranium-235 along the Radius of the Reactor.

The solid lines indicate the two-group two-dimensional calculation. The experimental values in the trap are multiplied by I.3. I - the trap is filled with water; 2 - 25 mm thick BeO plates are arranged in the trap along the core; 3 - Be inserts are arranged in the trap.

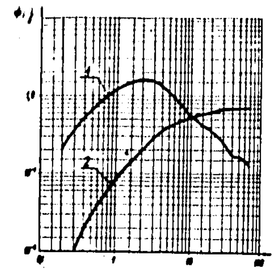

Fig.3. Energy Dependence of Neutron Flux ϕ (Z) and Flux along Energy Axis j(Z).
I - ϕ (Z); 2 - j(Z); $Z = \frac{E}{\kappa T}$;
T = 323°K.

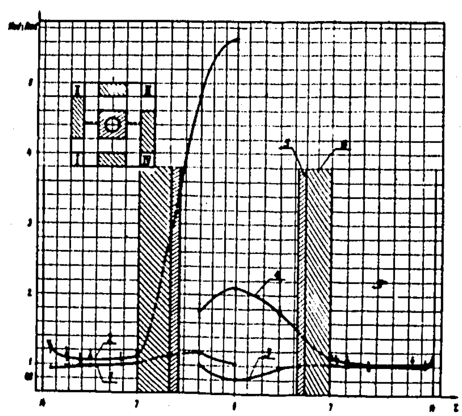

Fig.2. Experimental Distribution of Epicadmium Neutron Activation (Ncd) and Cadmium Ratio (Rcd) for Indium and Gold along the Radius of the Reactor.
I - N_{cd}^{Au} ; 2 - R_{cd}^{Au}' ; 3 - N_{cd}^{In} ; 4 - R_{cd}^{In} ;
5 - central channel wall; 6 - Be.

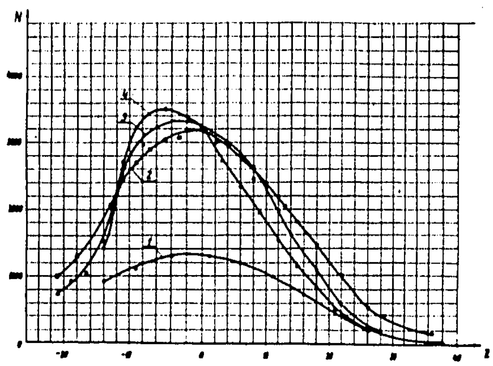

Fig. 4. Fission Density Distribution of Uranium-235 along the Height of the Vertical Channels I - channel No. I4; 2 - channel No. 4; 3 - channel No. I5 (the adjoining shim rod is taken out completely); 4 - channel No.I5 (the adjoining shim rod is taken out approximately by half).

320

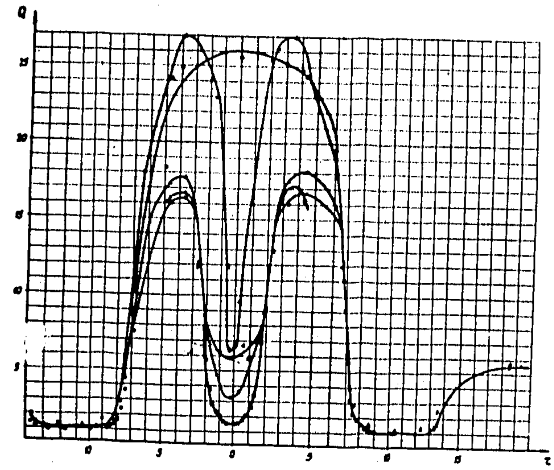

Fig.5. Fission Density Distribution of Uranium-235 in the Neutron Trap with the Fuel Plates of the Reactor [redacted]-2 Arranged in the Trap
⊖ - plates are absent; ▲ - 3 plates, spaced by 2.45 mm; ▲ - 6 plates spaced by 7.35 mm; x - 8 plates spaced by 4.9 mm; □ - I4 plates spaced by 2.45 mm

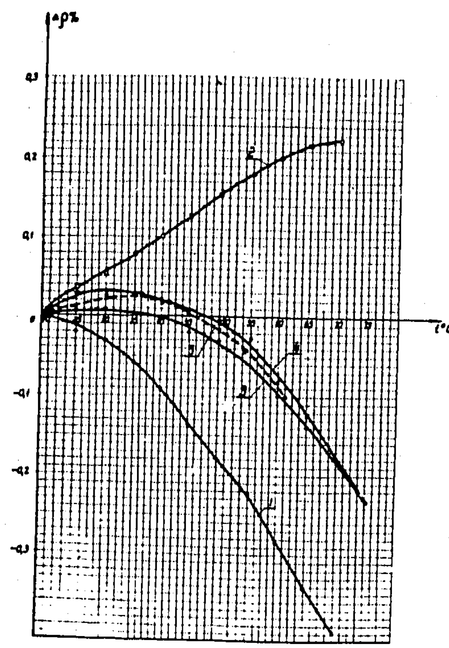

Fig.7. Temperature Effect of Reactivity
$\Delta\rho$ - variation of reactivity,%; t - water temperature, °C; I - heating of water in the main circuit; 2 - heating of water in channel No.I; 3 - simultaneous heating of water in the main circuit and in channel No.I; 4 - simultaneous cooling of water in the main circuit and channel No. I; 5 - simultaneous heating of water in the main circuit and channel No. I with a small volume of free gas present in the main circuit

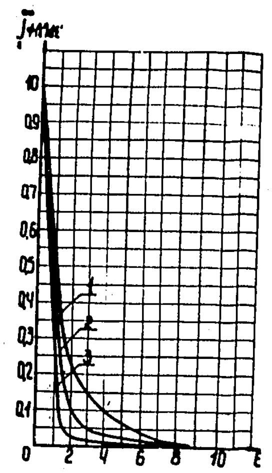

Fig.6. Integral Neutron Spectrum Measured by Threshold Indicators
E - neutron energy in Mev; I - spectrum in channel No. 4; 2 - spectrum in channel No. I9; 3 - spectrum in channel No.6

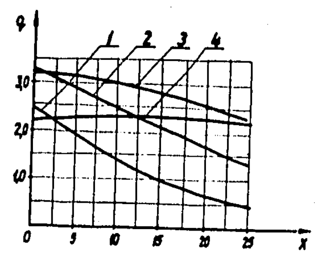

Fig.8. Variation of Thermal Load on the Surface of a Fuel Element versus Average Burnup of Uranium-235 in the Assembly (without Be Inserts).
q - the ratio of the thermal load on the given fuel element to the average thermal load in the assembly; x - average burnup of uranium-235,%;
I - end plate; 2 - second plate; 3 - third plate; 4 - fourth plate.

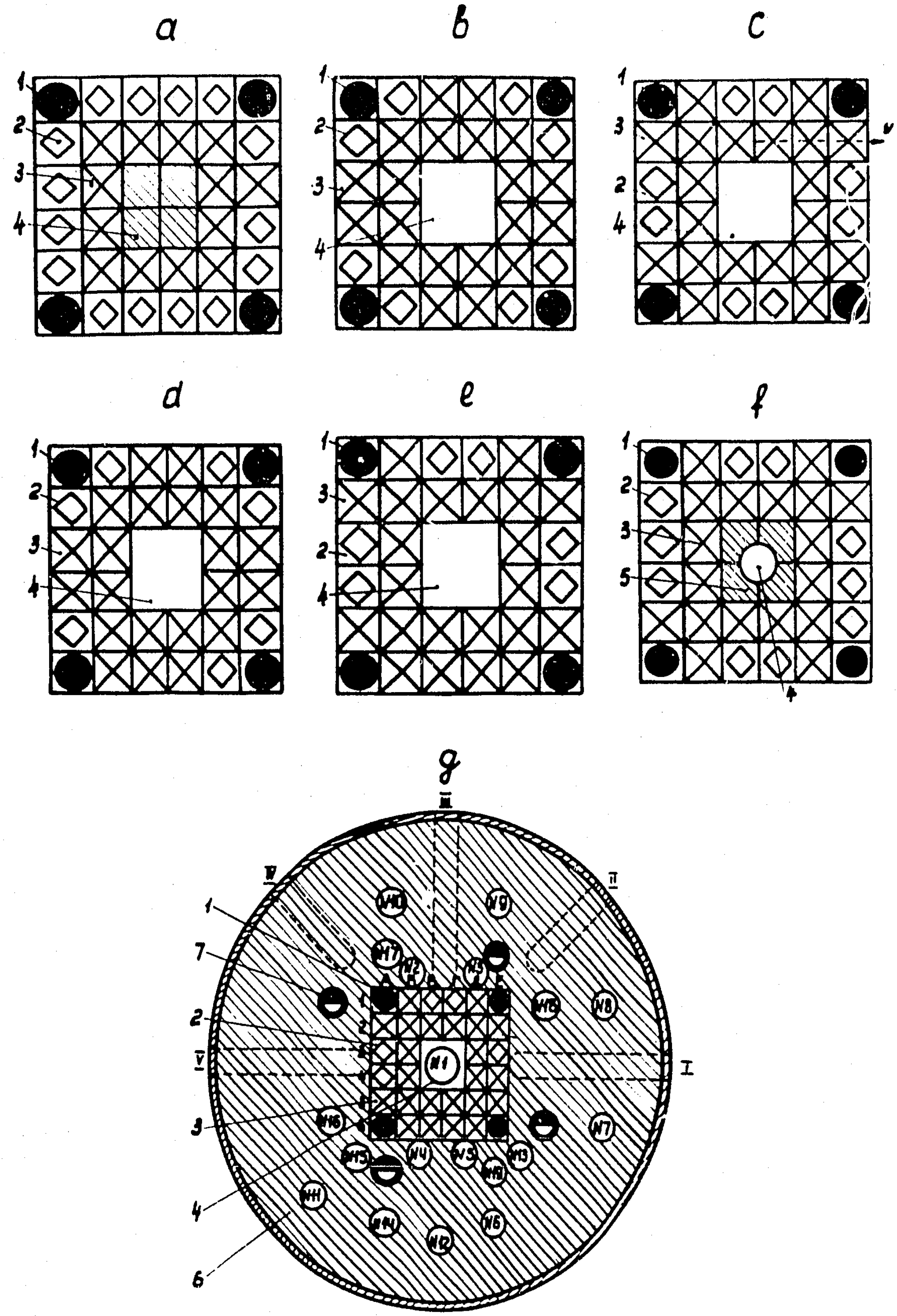

Fig.9. Cartograms.
I - shim rod; 2 - replaceable BeO assemblies; 3 - fuel assemblies; 4 - water; 5 - Be inserts; 6 - reflector; 7 - regulator rod.

320